§27. Investigation of Tritium Behavior and Tracability in In-Vessel Systems of LHD during D-D Burning

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Tritium (T) produced by D-D reactions is a safety concern in deuterium discharges planned in LHD. Estimation of tritium remained in and released from the vacuum vessel of LHD, chemical forms of released tritium by pumping, recovery of tritium, and how to reduced tritium release to the environment. As for tritium in the vessel, locations of high tritium retention and development of methods to remove tritium are

We have studied that tritium produced by DD-reaction behaves quite different way from hydrogen isotopes (hydrogen (H), and deuterium (D)) and even tritium (T) fueled into the tokamaks [1]. The results are summarized as,

(1) More than half or tritium produced by DD reaction would be implanted into plasma facing wall in the depth of more than 1□m, which can not be easily removed. Although high temperature thermal release of tritium implanted into the wall is promising, it can not be applied for tritium release from the whole vacuum vessel.

(2) Remaining tritium is thermalized to the plasma temperature and impinges the surface of the plasma facing wall. Most of them, however, would be replaced by subsequently coming fuel particles (H and D) and released with chemical forms of mostly DT or HT with little hydrocarbon and pumped out from the vacuum vessel.

(3) Tritium remaining in the vessel is very hard to quantize because various radiations from radio active nuclei produced by DD neutrons superpose tritium β electron, and should be discriminated for T analysis.

(4) In addition, the amounts of the retained tritium in the vessel must be strongly modified by temperature and materials of plasma facing tiles. The high energy tritium is implanted deeply with quite low concentration compared to the highly concentrated hydrogen co-depositions; the former is likely trapped strongly and hard to remove. If we could keep the temperature of the deposited area above 800 K, tritium inventory may be reduced significantly.

All above results are for tokamaks using carbon as a plasma facing material. In LHD, large part of PFM is composed of stainless steel, in which hydrogen solubility is less and hydrogen diffusivity is larger than those in carbon. Nevertheless, most of tritium having its initial energy of 1MeV is not thermalized in LHD plasma but implanted into plasma facing surfaces. Hence behavior of hydrogen implanted in stainless steels with rather high energy, up to 1MeV becomes important and should be investigated.

In addition, carbon materials eroded at divertor area could be transported and deposited on the first wall. Hence hydrogen behavior in not only bare stainless steels but also carbon deposition on them is our research targets.

Thus we have put particular emphases on,

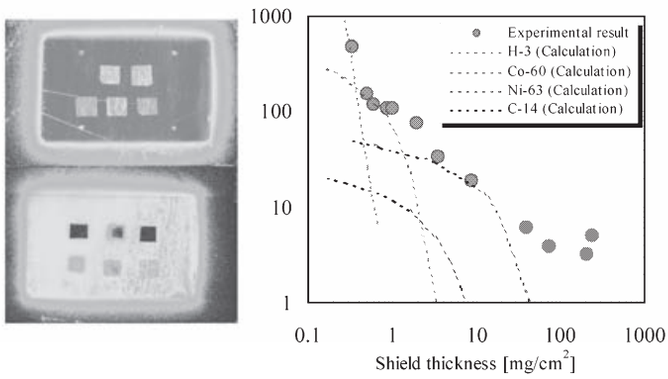

Fig.1 Discrimination of T from ^{60}Co, ^{63}Ni, and ^{14}C which were deposited on the bumper limiter of TFTR, using a film insertion method for radiation detection by an imaging plate technique [2]

1) To estimate tritium retention in the first wall and divertor area
2) To recover or remove diluted T in exhausted D gas during discharge.
3) To remove tritium implanted rather deep (more than 1μm) in the plasma facing wall.

In the preliminary study shows that above 400K, hydrogen retention in a stainless steel is very small and even long term release is possible. However, carbon deposition on the surface could enhance the tritium retention. This is very important for the ventilation of the LHD torous to keep people going into the vacuum vessel safe.

Concerning discrimination of tritium β-ray from other radio-activities made by DD neutrons we have tried a film insertion technique for tritium measurements by an imaging plate technique. Fig.1 is an example for discrimination of T from ^{60}Co, ^{63}Ni, and ^{14}C, which were deposited on the bumper limiter of TFTR [2].

For the purpose of tritium removal from carbon co-deposits on plasma-facing material of fusion device, pulsed-laser induced desorption of hydrogen from co-deposits on JT-60 open-divertor tile has been investigated. The fundamental (1064 nm) and fourth harmonic (266 nm) emission of a 20 ps-Nd: YAG laser were used, and dependences of hydrogen desorption on laser intensity IL and wavelength λ were studied. Hydrogen-desorption efficiency, defined as the ratio between the number of desorbed hydrogen by laser irradiation and that of hydrogen retained in the ablated volume, was largest in the region, where strong ionization of C^+ occurred, and was larger for λ = 266 nm, in which a laser photon can cut C-H bond, compared with that for λ= 1064 nm. For the ablative removal of hydrogen, a short-wave length and high-power laser is desirable [3].

References

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